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NUCLEAR ENERGY INSTITUTE

December 23, 2013

Mr. Michael E. Mayfield
Director, Advanced Reactor Program
Office of New Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: White Paper on Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone

Project Number: 689

Dear Mr. Mayfield:

Attached for NRC staff review and consideration is the subject white paper that the Nuclear Energy Institute (NEI)[1] has developed in response to SECY-11-0152.[2] The SECY notes the NRC staff's intent to develop a technology-neutral, dose-based, consequence-oriented emergency preparedness framework for small modular reactor (SMR) sites that takes into account the various designs, modularity and collocation, as well as the size of the emergency planning zone (EPZ), with the expectation that an applicant will provide a well-justified technical basis for NRC's review and consideration.

While the design-specific and site-specific technical basis will be provided by each SMR developer and applicants for a Combined Operating License and/or Early Site Permit under Part 52 or an Operating License under Part 50, the objective of this white paper is to propose a generic methodology and criteria that can be adopted and used for establishing the technical basis for SMR-appropriate EPZs. To that end, this paper is intended to serve as a vehicle to support the continuing dialogue with the NRC staff that should result in a mutually agreeable methodology and criteria, and thus provide the SMR developers and applicants sufficient guidance as they proceed to develop their design-specific and site-specific technical basis. This paper addresses SMRs with light water cooled and moderated designs only, and is not applicable to other types of

[1] The Nuclear Energy Institute (NEI) is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations and entities involved in the nuclear energy industry.
[2] U.S. NRC, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," SECY-11-0152, October 28, 2011.

SMRs or to large light water reactors. Also, the paper is limited to plume exposure EPZ. Ingestion exposure EPZ is to be addressed later.

Our proposed approach in the attached white paper is rooted in: (1) the expectation of enhanced safety inherent in the design of SMRs (e.g., increased safety margin, reduced risk, smaller and slower fission product accident release, and reduced potential for dose consequences to population in the vicinity of the plant); (2) the applicable SECY-11-0152 concepts including utilization of existing emergency preparedness regulatory framework and dose savings criteria of NUREG-0396[3]; and (3) the significant body of risk information available to inform the technical basis for SMR-appropriate EPZ, including severe accident information developed since NUREG-0396 was published in 1978, and information from the design-specific and plant-specific probabilistic risk assessments (PRAs) which will support SMR design and licensing.

Many of the key aspects of our approach were discussed with the NRC staff at a public meeting held on December 13, 2012. The approach addresses, among other things, the use of a suitable design-specific PRA and accounting for uncertainties important to establishing an appropriate SMR EPZ, and the need to address the effects of modularity and co-location. The attached paper is the first step for establishing the methodology and criteria, and it is expected that a series of increasingly more substantive, design-specific analytical reports implementing this methodology will follow.

While recognizing that NRC determination of an acceptable emergency preparedness plan for an SMR plant site will be made at the Combined Operating License Application or Operating License Application stage, the design certification applications and associated technical and/or topical reports by SMR vendors will contain a substantial amount of the technical information (e.g., source term, accident analyses, use of risk insights, and the role of enhanced plant features to address uncertainties) necessary to implement the methodology and criteria. Thus, establishing mutually acceptable methodology and criteria early via this white paper is important to support SMR design certification applications expected to be submitted beginning next year.

SECY-11-0152 discusses an approach in which the offsite EPZ is scaled to be commensurate with the SMR accident source term and associated dose characteristics, which are a function of the licensed reactor power level. The SECY indicates that such an approach to SMR EPZ sizing would: allow for regulatory predictability for SMR applicants and for State and local officials; ensure the consistent application of NRC regulations and requirements in the review of emergency preparedness plans prepared for SMRs; and, most importantly, be consistent with current emergency preparedness requirements and not result in a reduction in the protection of public health and safety. We believe that this EPZ approach for SMRs will have additional benefits as well. As a number of utilities are planning to retire many old, obsolete fossil plants within the next several years, the potential for SMRs as a viable repowering option has come into focus as a solution for fuel diversity and clean air considerations. For SMRs to replace many of these retiring fossil plants, an approach in which offsite EPZ is scaled, as noted above, will be a critical prerequisite because of these plants' location and site characteristics.

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[3] U.S. NRC, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396/EPA 520/1-78-016, December 1978.

We envision the technical basis established for determining SMR plume exposure EPZ size as one part of a comprehensive and integrated emergency preparedness plan that retains flexibility to accommodate differences in plant designs, projected accident source terms, and site characteristics, as appropriate. To that end, industry plans to initiate further dialogue with NRC regarding the ingestion pathway EPZ and developing generic SMR emergency preparedness plan guidance for addressing the 16 planning standards described in 10 CFR Part 50.47(b) and the associated requirements in 10 CFR 50, Appendix E. It is expected these additional efforts and the attached paper will be fully complementary.

We suggest a public meeting in the near future to receive NRC feedback on these matters and begin detailed discussion of the industry proposed approach for establishing the technical basis for SMR-appropriate EPZ. We will contact your staff shortly to identify a mutually convenient date for this meeting. We are prepared to continue the engagement with the NRC staff in the first part of calendar year 2014 to resolve comments and sustain progress toward establishment of a mutually agreeable methodology and criteria by mid-2014.

If you have any questions, please contact me or TJ Kim (tjk@nei.org; 202-739-8128).

Sincerely,

Douglas J. Walters

Douglas J. Walters

Attachment

c: Mr. Glenn M. Tracy, NRO, NRC
Mr. James T. Wiggins, NSIR, NRC
Mr. Robert J. Lewis, NSIR/DPR, NRC

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WHITE PAPER

PROPOSED METHODOLOGY AND CRITERIA FOR ESTABLISHING THE TECHNICAL BASIS FOR SMALL MODULAR REACTOR EMERGENCY PLANNING ZONE

December 23, 2013

ACKNOWLEDGMENT
This NEI White Paper was developed by the NEI Small Modular Reactor Licensing Task Force.

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Executive Summary

Emergency planning and preparedness is an essential element of a nuclear plant defense-in-depth strategy. NRC's SECY-11-0152[1] notes the NRC staff's intent to develop a technology-neutral, dose-based, consequence-oriented emergency preparedness framework for light-water small modular reactor (SMR) (also known as integral pressurized water reactor (iPWR)) sites that takes into account the various designs, modularity and co-location, as well as the size of the emergency planning zone (EPZ), with the expectation that an applicant will provide a well-justified technical basis for NRC's review and consideration. To that end, the objective of this white paper is to discuss a generic methodology and criteria that can be adopted and used by the SMR developers and plant operating license applicants for establishing the design-specific and site-specific technical basis for SMR-appropriate EPZs. This paper is intended to serve as a vehicle to support the continuing dialogue with the NRC staff that should result in a mutually agreeable methodology and criteria, and thus providing the SMR developers and applicants sufficient guidance as they proceed to develop their design-specific and site-specific technical basis.

This paper addresses SMRs with light water cooled and moderated designs only, and is not applicable to other types of SMRs or to large light water reactors. Also, the paper is limited to the consideration of plume exposure EPZ. Ingestion exposure EPZ is to be addressed later.

SECY-11-0152 recognizes that a scalable EPZ approach would: allow for regulatory predictability for SMR applicants and for State and local officials; ensure the consistent application of NRC regulations and requirements in the review of emergency plans prepared for SMRs; and, most importantly, be consistent with the objectives of current emergency preparedness requirements and not result in a reduction in the protection of public health and safety. Industry believes that siting and building SMRs with appropriate EPZ size will have additional benefits as well. As a number of utilities are planning to retire many old, obsolete fossil plants within the next several years, the potential for SMRs as a viable repowering option has come into focus as a solution for fuel diversity and clean air considerations. For SMRs to replace many of these retiring fossil plants, an appropriately-sized EPZ will be a critical prerequisite because of these plants' location and site characteristics.

The proposed industry approach is rooted in: (1) the expectation of enhanced safety inherent in the design of SMRs (e.g., increased safety margin, reduced risk, smaller and slower fission product accident release, and reduced potential for dose consequences to population in the vicinity of the plant); (2) the applicable SECY-11-0152 concepts including utilization of existing emergency preparedness regulatory

[1] U.S. NRC, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," SECY-11-0152, October 28, 2011.

framework and dose savings criteria of NUREG-0396[2]; and (3) the significant body of risk information available to inform the technical basis for SMR-appropriate EPZ, including severe accident information developed since NUREG-0396 was published in 1978, and information from the design-specific and plant-specific probabilistic risk assessments (PRAs) which will support SMR design and licensing.

Industry agrees with the SECY concepts of a dose/distance approach to establish the EPZ boundary for light water-SMRs, the notion that emergency plan requirements for SMRs should be commensurate with the accident source term and associated dose characteristics for the designs, and the use of a sizing rationale that is analogous to that in NUREG-0396. These concepts, as well as potential impacts on reactor modules from common or shared systems, effects of co-location, and steps to reflect lessons learned from the Fukushima accident are reflected in the methodology and criteria proposed in this paper.

The SECY also notes, in considerations for establishing an appropriate EPZ size for SMR, that it is industry's responsibility to develop and implement the detailed methodology and criteria for review and approval by the staff including: (1) addressing the use of a suitable design-specific PRA; and (2) accounting for uncertainties. The proposed methodology for establishing the technical basis uses a risk-informed approach with two complementary efforts which address the two SECY considerations:

1. Use of information from the PRA required for new plant designs to inform accident sequence selection, determine release timing and release magnitude, and determining offsite doses; and

2. Enhanced plant capabilities to account for uncertainties, including an operationally-focused mitigation capability and other features emphasizing traditional engineering insights.

The methodology in this white paper has been developed to learn from industry experience with risk-informed decision making on regulatory applications. It is intended that the methodology be part of an integrated, decision-making process for SMR EPZ sizing which uses risk informed judgment in which insights from PRA are considered together with other engineering insights. The goal is that a balance be achieved between use of PRA information and application of a deterministic, defense-in-depth perspective.

Use of the PRA to Inform EPZ

The design-specific and plant-specific PRA will be used to implement the NUREG-0396 framework to the SMR EPZ sizing approach including dose savings objectives and consideration of a spectrum of accidents. This part of the methodology is quantitative and uses the significant body of risk information noted above. Proposed technical criteria for determining SMR EPZ size, taken from NUREG-0396 are as follows:

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[2] U.S. NRC, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396/EPA 520/1-78-016, December 1978.

a. The EPZ should encompass those areas in which projected dose from design basis accidents (DBAs) could exceed the EPA protective action guides (PAGs)

b. The EPZ should encompass those areas in which consequences of less severe core melt accidents could exceed the PAGs

c. The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe core melt accidents

The three criteria are to be addressed for a given SMR design by: using radiological DBA information from Chapter 15 of the Safety Analysis Report (Criterion a); applying PRA Level 1 information to define and inform the selection of more probable (Criterion b) and less probable (Criterion c) accident sequences, where the less probable accident sequences will include extreme external hazards and the potential impact on reactor modules that have some common or shared systems; applying Level 2 PRA information to define fission product releases for these sequences (Criteria b and c); and performing offsite dose calculations to determine the dose associated with these sequences. The offsite dose results are then compared against the applicable criteria from a, b, and c above to provide input to determination of an appropriate EPZ for a design.

Steps to Account for Uncertainties

Steps to account for uncertainties have been defined in parallel with and as a complement to applying information and insights from the PRA. These steps are more qualitative and involve enhanced plant capabilities intended to address and compensate for uncertainties in PRA results (accident initiation, plant response, and accident progression) and matters which cannot be treated in the PRA. Key aspects of these enhanced plant capabilities are as follows:

- Provision of additional severe accident mitigation capability (beyond the installed plant systems and structures) that is operationally-focused. This mitigation capability will not be based on specific accident sequences nor on probabilities, but rather on maintaining basic safety functions in the face of extreme site-wide events and situations where it is difficult to foresee all potential conditions in advance.

- Assessing potential risks that are difficult to quantify or not fully addressed in the PRA such as security events, limited operating experience, co-location, or risks for which industry standards may not be available. In this regard, applicants should perform an engineering assessment (qualitative or quantitative as appropriate) either to show that a given risk is adequately treated inside the PRA; or to confirm the existence, functionality, and capability of features and processes in the design, operation, accident management, and emergency response to address this risk and to provide confidence that the risk impact is acceptably low.

- Assessing the potential impact on risk of very low frequency accident sequences in the PRA, beyond the set of sequences utilized in the PRA-based part of the methodology. This can be accomplished by assessing the potential for cliff-edge effects from these low frequency sequences, and/or from historically important higher consequence accident scenarios in past LWR PRAs, which could cause the risk to significantly exceed that from the selected accident sequences. Development of the operationally-focused mitigation capability noted above, where this capability is based on maintaining basic safety functions regardless of the threat, provides an alternative for addressing such low frequency events when quantitative risk evaluations and PRA treatment are impractical. This mitigates the need to resort to arbitrary and/or extreme, overly conservative scenarios as the basis of EPZ sizing.

- Provide reasonable balance between prevention, mitigation, and protective actions (an essential element of defense-in-depth). To accomplish this, both an onsite and an offsite emergency plan would be provided, including a certified offsite all hazards plan. To that end, NEI plans to initiate further dialogue with NRC regarding the ingestion pathway EPZ and developing generic SMR emergency plan guidance for addressing the 16 planning standards described in 10 CFR 50.47(b) and the associated requirements in 10 CFR Part 50, Appendix E. It is expected that these additional efforts and this paper will be fully complementary.

Future Steps to Develop SMR EP

While recognizing that NRC determination of an acceptable emergency plan for an SMR plant site will be made at the Combined Operating License Application or Operating License Application stage, industry believes the design certification application and associated technical and/or topical reports by SMR vendors will contain a substantial amount of the technical information (e.g., source term, accident analyses, use of risk insights, and the role of enhanced plant features to address uncertainties) necessary to implement the methodology and criteria. Thus, establishing an acceptable methodology and criteria early via this white paper is essential to support SMR design certification applications expected to be submitted beginning in 2014.

Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone

1. Introduction and Scope

The objective of this white paper is to describe a proposed methodology and criteria for establishing the technical basis associated with small modular reactor (SMR) emergency planning zone (EPZ) sizing. The paper is in support of the continuing dialogue with NRC on emergency preparedness (EP) and SMR-appropriate EPZs, and responds to SECY-11-0152 [1], "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," which discusses the NRC staff's intent to develop an EP framework for SMRs. The paper addresses SMRs with light water cooled and moderated designs only, and is not applicable to other types of SMRs nor to large light water reactors (LWRs).
The technical basis for determining the EPZ size which is appropriate for SMRs is rooted in their enhanced safety. This technical basis recognizes and allows for what is expected to be reduced risk and increased safety margins of the SMR designs, including smaller cores and smaller, slower fission product releases in an accident.

At a high level, the paper is a first step in developing a methodology for establishing the technical basis for determining EPZ size. It proposes a risk-informed approach with two complementary efforts: (1) using the probabilistic risk assessment (PRA) required for new plant designs to inform EPZ sizing considerations, and (2) providing enhanced plant capabilities to account for uncertainties, including an operationally-focused mitigation capability.

The PRA would be used to apply the NUREG-0396 [2] framework to the SMR EPZ sizing approach including dose savings objectives and consideration of a spectrum of accidents. This part of the methodology is quantitative and uses the significant body of risk information expected to be available to inform the technical basis for EPZ sizing including LWR severe accident information developed since NUREG-0396 and information from detailed SMR design-specific and plant-specific PRAs which will support SMR design and licensing.

Providing enhanced plant capabilities, in parallel with and as a complement to the PRA-based evaluation, is more qualitative and is intended to address and compensate for uncertainties in PRA results (accident initiation, plant response, and accident progression) and matters which cannot be treated in the PRA. A key aspect of the enhanced plant capabilities is provision of an additional accident mitigation capability (beyond the installed plant systems and structures) that is operationally-focused and not based on specific accident sequences nor on probabilities but rather on maintaining basic safety functions in the face of extreme site-wide events and situations where it is difficult to foresee all potential conditions in advance.

Use of PRA is essential in the proposed methodology, and is the means to address the SECY-11-0152 provisions that: the dose/distance approach be applied to establish the SMR EPZ boundary; and EP requirements should be scalable for SMRs as illustrated in Table 1 of the SECY. At the same time, industry experience shows that attempts at applying quantitative, PRA-based information in decision-making on regulatory matters have proven difficult. Uncertainties associated with state of knowledge limitations and with hazards and events not easily amenable to PRA are hard to deal with in practice and often lead to overly conservative, extreme solutions. The enhanced plant capabilities in the methodology addresses this by: providing a balance between deterministic, defense-in-depth considerations and risk considerations; providing an alternative in the face of very low frequency events or other matters where quantitative risk evaluations and analytic treatment of uncertainties are impractical; and mitigating the need to resort to arbitrary and extreme scenarios as the basis of EPZ sizing.

Development of SMR EP planning standards and providing a substantial base for expansion of response, though mentioned in this white paper to provide context for the EPZ effort, are not addressed in the paper. They will be addressed in future industry submittals as part of an integrated treatment of EPZ sizing methodology and EP planning requirements appropriate to SMRs. Ingestion pathway EPZ will be addressed as progress is made on agreement on the approach for sizing of the plume exposure EPZ.

Section 2 of the paper briefly describes the reasons for addressing EPZ size for SMRs and its relevance and importance to all stakeholders. Section 3 contains the PRA-based evaluation portion of the proposed methodology and criteria. The proposed methodology and criteria address LWR SMRs (also known as integral pressurized water reactor (iPWR) designs) and address both single module and multi-module designs. Section 4 contains additional steps in the methodology to account for uncertainties, i.e., key matters associated with the enhanced plant capabilities. Section 5 proposes next steps for NRC - industry interaction on SMR EPZ and a path forward to address EPZ sizing.

SECY-11-0152 discusses the NRC staff's intent to develop a technology-neutral, dose-based, consequence-oriented EP framework for SMRs. In describing an approach for scalable EPZs, the SECY notes:

> "The staff has reviewed the existing EP requirements associated with various nuclear facilities and has identified that all of the existing types of NRC-licensed nuclear facilities use a dose/distance approach to establish the boundary of their EPZ (or other planning area) based on the Environmental Protection Agency Protective Action Guidelines. The staff concluded that a similar technology-neutral dose/distance rationale would also be appropriate for the advanced designs.
>
> The approach the staff is developing is based on the concept that EP requirements could be scaled to be commensurate with the accident source term, fission product release, and

> associated dose characteristics for the designs. As the staff is developing the approach, issues related to modularity of the designs and the potential for collocating the reactors near industrial facilities are also being explored.
>
> …the staff recognizes the need to reflect in the anticipated framework, the lessons learned at the conclusion of agency task force reviews from the accident at the Fukushima Dai-ichi nuclear power plant in Japan."

Industry agrees with the dose/distance approach to establish the EPZ boundary and the concept that EP requirements should be scalable for SMRs commensurate with the characteristics of their fission product release during a postulated accident. These matters are reflected in the design-specific, PRA-based evaluation discussed in Section 3 of this white paper. Industry also understands the need to explore issues related to modularity of the designs and the potential for co-locating the reactors near industrial facilities. These issues are reflected in the Section 3 PRA evaluation and the Section 4 additional steps to account for uncertainties. Finally, industry agrees with the need to reflect, as appropriate, lessons learned at the conclusion of agency task force reviews from the Fukushima accident. The white paper is a first step to reflect these lessons.

SECY-11-0152 also states the following with regard to considerations for establishing the size of EPZs for SMRs:

> "The staff anticipates drawing on the substantial improvements over the last several years in understanding and modeling of severe accident phenomena. The staff anticipates that an appropriate method for use in this application would involve (1) using a PRA …to calculate the probability of exceeding PAG as function of distance from the exclusion area boundary for a spectrum of accidents, (2) establishing criteria for determining the point at which the probability of exceeding the PAG is acceptably low, and (3) concluding that the events provide an acceptable spectrum of consequences. Although a more rigorous design and site-specific approach, the staff anticipates that this approach will be generally analogous to that discussed in NUREG-0396.
>
> …it is anticipated that the industry will develop and implement the detailed calculation method for review and approval by the staff. The staff acknowledges a number of challenges in implementing the approach, such as developing a suitable SMR design-specific PRA and accounting for the uncertainties in the state of knowledge of SMR designs. The staff will continue to work with stakeholders on this issue."

Industry understands and agrees with the SECY-11-0152 statement regarding the considerations for establishing SMR EPZ size, and that it is industry's responsibility to develop and implement the detailed calculation method and criteria for review and approval by the staff including addressing the challenges of a suitable design-specific PRA and accounting for uncertainties (hence the two complementary aspects of the methodology in Sections 3 and 4). This paper is provided as a first step in this direction.

Additional key aspects of the industry-proposed approach for establishing the SMR EPZ size technical basis are as follows:

- The demonstration of the technical basis for EPZ size will be made on a design-specific basis with the burden on the designer to make the technical case relative to the design and on the combined operating license (COL)/early site permit (ESP) applicant (or an applicant under Part 50) with regard to the site. Industry has prepared this paper on SMR EPZ sizing at this point in time in response to SECY-11-0152 and since it is anticipated that a significant amount of the information necessary to implement the technical basis for a given design will be included in the safety analysis report (SAR) which supports licensing.
- The technical basis for an appropriately sized EPZ should be part of a comprehensive approach for EP response and protective actions that retains flexibility to accommodate differences in plant designs, projected source terms, and possibly site characteristics. The SMR EP framework conceptualized in SECY-11-0152 and described herein represents an evolution relative to the approach implemented under current EP regulations in which no licensee analysis is required for establishing EPZ size.

2. Reasons to Consider Appropriate EPZ Sizing for SMRs

Emergency planning for protective actions within zones around a nuclear power plant has been an NRC requirement since the early 1960's. Initially, 10 CFR Part 100 required that every site must have an exclusion area and a low population zone (LPZ). Later, a joint NRC/EPA Task Force was chartered, the goal of which was to provide more definitive, clarifying guidance. The regulatory framework defining specific EPZ sizes for power reactors was an outcome of this joint Task Force effort, published in 1978 as NUREG-0396. In 1979, the NRC issued a policy statement describing two EP planning zones: a plume exposure EPZ of about 10 miles and an ingestion pathway EPZ of about 50 miles. The plume EPZ is for detailed planning and rapid response, and provides a base for expansion beyond the EPZ boundary if necessary. The ingestion EPZ is for longer term actions.

Following the Three Mile Island accident, these two EPZs were included in the 1980 rulemaking that added 10 CFR § 50.47 as a means to establish standards upon which emergency plans are to be reviewed. As the size requirements for the two EPZs were derived from conservative analyses for large LWR plants, the 1980 rulemaking allowed that the size of the EPZs may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. For these plants, smaller EPZs were permitted due to their reduced risk. Appendix A provides more detail on the regulatory background.

Sizing requirements for EPZs, stated in 10 CFR § 50.47, are based on conservative analyses for large LWRs contained in WASH-1400 [3]. Insights from 50 plus years of industry design and operating experience are now available, together with growth of the experimental data base on radionuclide release during an accident and the analytical tools available to calculate such releases in the nearly four decades since the WASH-1400 report was published in 1975. Additionally, NRC's policy statements on safety goals, severe accidents, advanced reactors, and the use of probabilistic risk assessment (PRA) have greatly encouraged advancements and innovations in plant designs and safety evaluation methods and criteria.

An SMR that is located at a greenfield site, co-located at a site with industrial customers, or repowering an existing fossil site presents an unique situation. For SMRs the benefits of appropriate EPZ sizing are significant. SMRs hold significant promise in meeting energy needs worldwide for: inherently safe, scalable, economical electric power generation; electric power generation at a distance from large grid systems; and applications in addition to electric power generation such as water desalination and process heat. Successful development and deployment of these new technologies requires commensurate and timely regulatory evolution, including in the area of EP.

There are several reasons for reconsidering EPZ sizing for SMRs. First, the SMR designs are different from traditional, large LWR plants in ways which significantly reduce the potential for offsite fission product release and dose consequences (e.g., smaller core fission product inventories, improved design features, slower accident sequence evolution). The EPZ size for SMRs should reflect their design, source terms, and severe accident dose characteristics. Second, there have been significant advancements over the last several decades in the understanding of severe accidents, fission product release and transport phenomena, consequence analysis, and effectiveness of offsite protective actions, all of which suggests smaller, slower fission product releases during accidents and reduced health and safety risks to the public as compared with earlier conservative analyses. Third is that implementation of appropriate EPZ sizing can simplify interfaces between the plant operator, the surrounding communities, and any co-located customers. This benefits both the communities and the licensee, and will significantly contribute to successful deployment of SMRs in the U.S.

Industry believes that siting and building SMRs with appropriate EPZ size and planning elements will have benefits for all stakeholders. This is based on the expectation that the SMR overall safety case and defense-in-depth, including design, operation, security, and appropriate EPZ and planning elements, will further enhance the design and safety margins and further reduce accident risk to the public. Table 1 summarizes the various stakeholders and anticipated benefits of siting and building such SMRs.

Table 1 Summary of Benefits to SMR Stakeholders

Stakeholder	Benefit to Stakeholder of Siting and Building SMRs with Appropriate EPZ Size and Planning Elements
State and local offsite agencies	Optimizes utilization of resources, simplifies and improves coordination of emergency response (potentially smaller area, fewer jurisdictions involved in response)
Licensees	Increased siting possibilities, better focus of resources for public health and safety protection, better control of risks and costs
Public in vicinity of plant	No reduction in protection of public health and safety, reduced overall health risks, reduced population subjected to unnecessary disruption associated with potential evacuation
Co-located customers	Minimizes impact on customer facility operation and associated emergency response plan, provides opportunity for consistent EP response as part of National Response Framework (NRF)
Regulators (NRC, FEMA)	More up-to-date, transparent EPZ sizing basis
Department of Homeland Security	Facilitates integration of nuclear plant emergency response into NRF
Public-at-large	Societal benefits from deployment of SMRs (infrastructure development, jobs, economic development, grid use, land use, reduced greenhouse gas emissions, etc.)

3. PRA-Based Evaluation

Section 3 discusses the PRA-based evaluation portion of the methodology and criteria. Section 4 discusses additional steps to account for uncertainties. Key aspects of the PRA-based evaluation are as follows:

- It is proposed that the PRA-based evaluation portion of the SMR EPZ size methodology be evolutionary in the sense that it would maintain the framework and dose savings objective from NUREG-0396, and also draw on the EPRI ALWR EPZ study [4] and peer review [5].[1] At the same time, the approach relies on the substantially greater severe accident and PRA knowledge base available today.

[1]The NRC supports maintaining the NUREG-0396 approach, as indicated in SECY 97-020 and SECY-11-0152 which state that the current rationale for the size of the EPZ, i.e., potential consequences from a "spectrum of accidents" tempered by probability considerations, should be maintained.

- The PRA-based evaluation should consider risks that are amenable to a PRA treatment. PRAs used in risk-informed applications may vary in scope and level of detail depending upon the application and the plant stage (design, construction, or operation). For the PRA-based evaluation to support the SMR EPZ size technical basis, Level 1 and Level 2 PRA should be completed.

- The following attributes need to be addressed:
  - Applicable plant operating states (modes) including full power, and low power and shutdown (LPSD). There may also be design-specific operating states unique to certain modular SMR designs which need to be addressed involving, for example, refueling and/or concurrent power operation and refueling.
  - Accident initiators amenable to PRA including internal hazards and external hazards.
  - In addition to core damage, fuel handling accidents (FHAs), and spent fuel pool (SFP) accidents
  - Core damage frequency (CDF) and large early release frequency (LERF)

- A Level 3 PRA is not required, but offsite dose calculations will be necessary. NRC-endorsed PRA standards are now available only for Level 1 and limited Level 2 internal events and external events for reactor accidents at power. Full Level 2 and LPSD standards are being developed. For operating states, initiators, or accidents for which standards have not yet been endorsed, applicants may address the associated risks by margins type approaches or other systematic evaluation techniques per Section 4.2. References [6] and [7] provide further details on PRA scope.

- As is the case for all new design plant PRAs, the level of detail will progressively evolve with the plant stage.

- NUREG-0396 and SECY-97-020 [8] indicate that the margins of safety provided by the 10-mile EPZ for existing plants were not based on quantification of accidents, but rather "were qualitatively found adequate as a matter of judgment" [8]. This concept for determining the adequacy of the margins of safety needs to be updated to include risk-informed judgment as part of an integrated decision-making process for SMR EPZ sizing as discussed in Section 3.1. In the over three decades since NUREG-0396 was published, the severe accident experimental knowledge base and analytical methods have advanced to the point that tools and models are now available to support a risk-informed approach to justification of EPZ size and the associated safety margins for SMRs. The fully integrated, engineering level, severe accident analysis computer code, MELCOR [9], industry's MAAP5 code, and the NRC's State-of-the-Art Reactor Consequence Analysis (SOARCA) study [10] are examples of usage of advanced tools and models.

3.1 SMR EPZ Sizing Rationale and Use of Risk-Information

The primary objective of EP as indicated in NUREG-0396 and restated in references [1] and [8] is to produce dose savings for a spectrum of accidents that could potentially lead to offsite doses in excess of the EPA protective action guides (PAGs) [11]. This is a reasonable objective that is considered applicable to LWR SMRs and is consistent with the EPZ sizing rationale proposed herein.

Industry is proposing to apply the NUREG-0396 sizing rationale to SMR EPZ size determination, and at the same time to apply the significant body of severe accident information that has been developed in the over three decades since NUREG-0396 was published, and to apply the design-specific and plant-specific PRA information that will be prepared to support SMR licensing.

NRC has been moving increasingly in a direction of applying risk information to regulatory matters for over a decade as discussed in Appendix A. References [12], [13], [14], and [15] are good examples of this. The NRC Near Term Task Force (NTTF) Review of Insights from the Fukushima Dai-ichi Accident [16] proposed, in Recommendation 1, establishing a "logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations". NRC has work ongoing [17] to evaluate and disposition this recommendation including consideration of a new category of beyond design basis events and use of risk insights and plant-specific PRAs where appropriate as part of event identification and categorization.

The industry proposed use of risk information in EPZ size determination is not a risk-based approach. Rather, it is to inform, i.e., a risk-informed approach, as discussed in reference [18]. As shown in Figure 1, a risk-informed approach is a combination of traditional and risk-based approaches through a deliberative process. It balances risk considerations and defense-in-depth.

Two good examples of a deliberative process for incorporating risk insights into decision-making are Regulatory Guide 1.174 [12] and the recent NRC report, "A Proposed Risk Management Regulatory Framework" (RMRF) [13]. Regulatory Guide 1.174 specifies use of PRA methods and data in a manner that complements the NRC's deterministic approach and indicates NRC's desire to base its decisions on the results of traditional engineering evaluations, supported by risk insights. Regulatory Guide 1.174 also describes principles of risk-informed, integrated decision-making that include addressing defense-in-depth and maintaining safety margin in parallel with use of risk analysis techniques. These principles, originally defined in 1998 to address use of PRA in risk-informed decisions on plant-specific changes to the licensing basis, are illustrated in Figure 2 of reference [12], repeated as Figure 2 below, and are relevant and apply to the decision-making process for any risk-informed application.

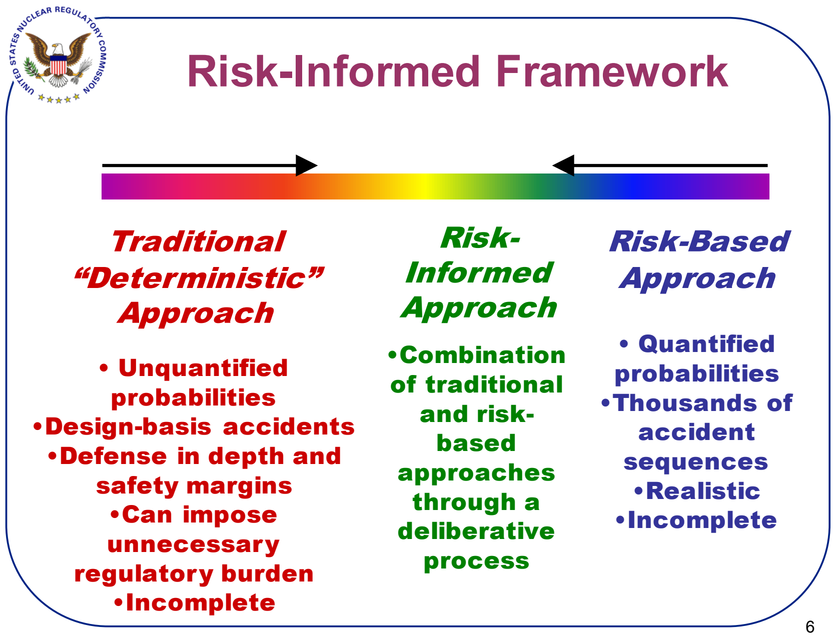

Figure 1. Risk Informed Framework (reproduced from reference [18])

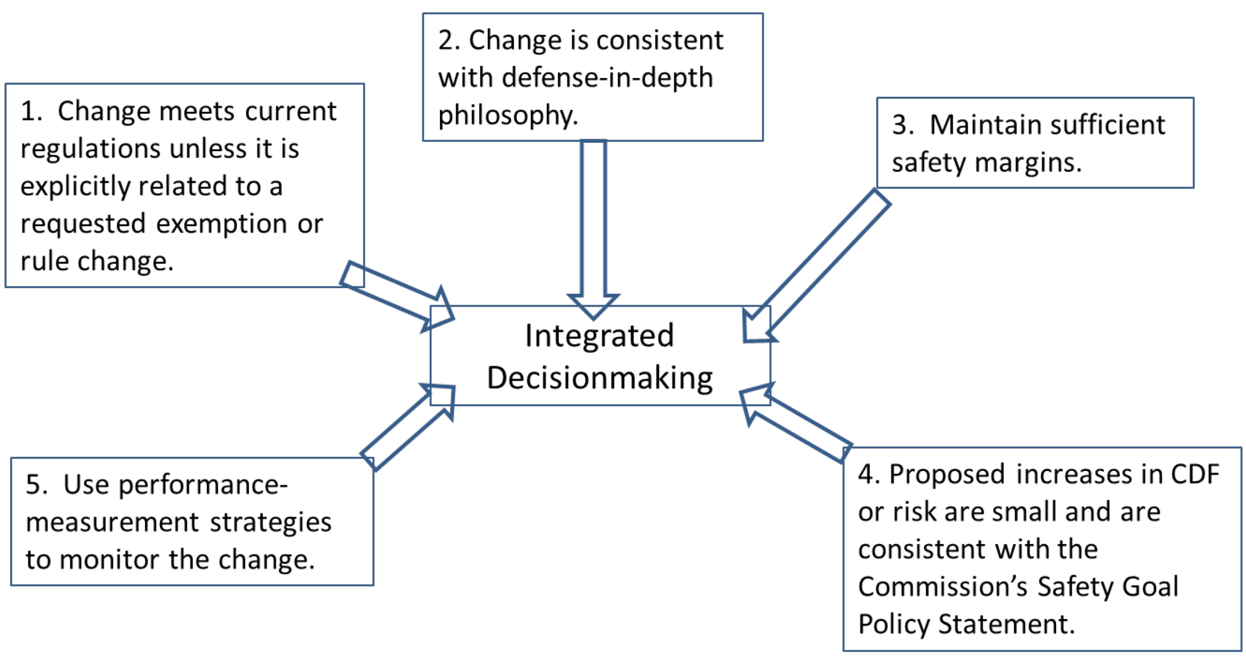

Figure 2 Principles of Risk-Informed Integrated Decision-Making (taken from reference [12])

The recently proposed NRC RMRF suggests use of a disciplined decision-making process as part of the RMRF, as shown in Figure 2-1 of reference [13], repeated as Figure 3 below, to achieve risk management goals.

Figure 3. Decision-Making Process as Part of a Risk Management Regulatory Framework (taken from reference [13])

Though not discussed further in this white paper, industry envisions that the use of risk information as part of the decision on SMR EPZ size should be part of a deliberative, integrated process (integrating risk results, defense-in-depth considerations, and other factors) similar to the examples above, and that the EPZ size decision should be made in context with decisions on the SMR planning standards and confirmation of substantial base for expansion of response.

It is intended that the methodology proposed herein be part of an integrated, decision-making process for SMR EPZ sizing which uses risk informed judgment in which insights from PRA are considered together with other engineering insights. The goal is that a balance be achieved between use of PRA information and application of a deterministic, defense-in-depth perspective, such that the technical basis for EPZ size is insights, not just numbers or criteria. As described in the remainder of Section 3 and in Section 4, the methodology uses probability information as input to selection of accident scenarios to be evaluated against the dose-based criteria so as to provide a means to support the credibility of the accident scenarios. At the same time, the methodology maintains the part of the existing sizing rationale on addressing a spectrum of accidents, including assessing the potential for cliff-edge effects[2] and other steps to account for uncertainties.

3.2 SMR EPZ Sizing Technical Criteria

The technical criteria, which flowed from the NUREG-0396 sizing rationale and were used to determine the generic distance for the plume exposure EPZ for existing plants, were stated in reference [8] and are as follows:

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[2] Cliff-edge effects is a term used in the nuclear safety community to refer to events or faults for which a small incremental decrease in frequency can yield a disproportionate increase in consequences. See references [19] and [20] which refer to cliff-edge effects.

d. The EPZ should encompass those areas in which projected dose from design basis accidents (DBAs) could exceed the PAGs
e. The EPZ should encompass those areas in which consequences of less severe core melt accidents could exceed the PAGs
f. The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe core melt accidents

Reference [8] also stated that "detailed planning within the EPZ was expected to provide a substantial base for expanding response efforts should expansion be necessary for those low probability, high consequence events whose effects could extend beyond the EPZ." Industry considers these technical criteria to be appropriate for SMR EPZ sizing.

Figure 4 illustrates an Integrated SMR EP Approach which includes the technical criteria for EPZ sizing and also addresses additional steps to account for uncertainties. There are five parts to the approach as shown in Figure 4:

Part i Addresses technical criteria a. and b., both of which involve more probable, less severe accidents and comparison of consequences against the EPA PAGs
Part ii Addresses technical criterion c. which involves less probable, more severe accidents and comparison of consequences against early severe health effects
Part iii Addresses additional steps to account for uncertainties
Part iv Develops the SMR emergency planning elements which are informed by the results of Parts i, ii, and iii
Part v Confirms that the emergency plan provides a substantial base for expansion of response if necessary

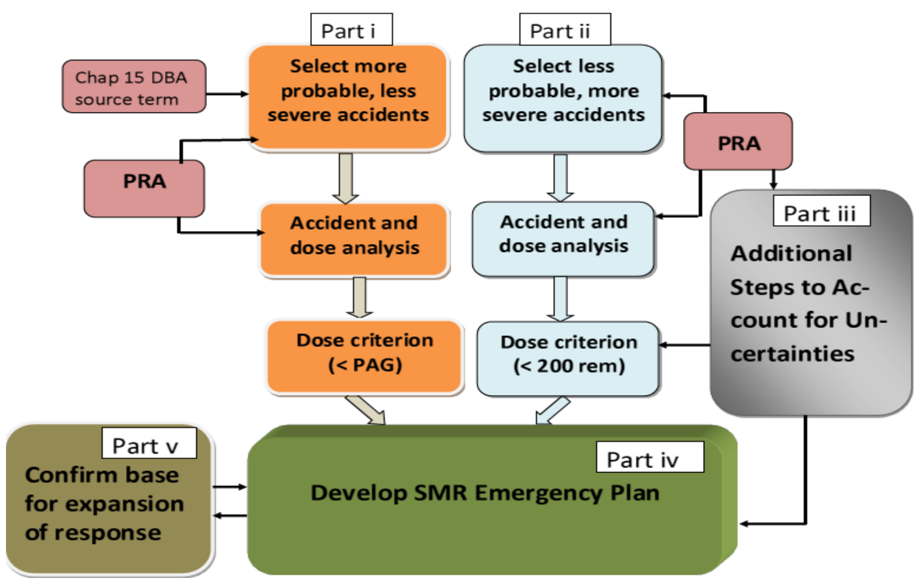

Figure 4 Integrated SMR EP Approach

This paper addresses only Parts i, ii, and iii since the methodology for these parts must be developed prior to addressing Parts iv and v. Parts iv and v will be addressed in the future, in concert with the EPZ size decision and using risk-informed methods, after progress is made on Parts i, ii, and iii.

3.3 Methodology for Implementing Criterion a: "The EPZ should encompass those areas in which projected dose from DBAs could exceed the PAGs" (Part i)

Elements of the methodology for implementing Criterion a. include:

- Accident scenario selection (top orange box on Figure 4)
- Accident scenario source term evaluation methodology where source term in this context refers to fission product release to the environment as a function of time (second orange box on Figure 4)
- EPZ boundary consequence calculation methodology (also second orange box on Figure 4); the relevant consequence parameter is dose
- Comparison of dose and consequences with the PAGs (third orange box on Figure 4)

Each of these elements is discussed below.

Accident Scenario Selection: For Criterion a. accident scenario selection is relatively simple in that the most challenging DBA source term used for the SAR Chapter 15 analysis will be applied (top left lavender box on Figure 4). The "most challenging" DBA source term has traditionally been the Regulatory Guide 1.183 [21] loss of coolant accident (LOCA) source term for operating plants and large ALWRs although, due to design differences in SMRs versus large plants, the most challenging Chapter 15 source term for a given SMR design may have deviations from Regulatory Guide 1.183. Industry's position paper on "Small Modular Reactor Source Terms" [22] discusses two approaches to establishing an accident source term for SMRs: one based upon the Regulatory Guide methodology and a second based upon a new methodology that takes into consideration specific SMR design characteristics.[3] This Chapter 15 accident is likely to be the limiting DBA from a dose standpoint though this will need to be confirmed by the applicant.

Accident Source Term Evaluation: The accident source term evaluation methodology for Criterion a. will be simply to apply the results of the analysis of fission product release to the environment versus time which must be performed as part of the SAR and can be extracted from Chapter 15. Thus, little or no additional work is expected to be necessary for either accident scenario selection or accident source term evaluation methodology in conjunction with implementing Criterion a.

[3] As stated in the introduction to Section 3.3, "source term" in the context of the EPZ sizing accident sequence source term evaluation methodology refers to fission product release to the environment as a function of time. The Regulatory Guide 1.183 source term, on the other hand, is the fission product release into containment. Additional analysis on the part of the applicant is then necessary as part of Chapter 15 to evaluate fission product transport in containment and fission product leakage from containment to the environment.

EPZ Boundary Consequence Evaluation: The EPZ boundary consequence calculation methodology should apply a methodology similar to that used in the SOARCA study [10] which used state-of-the-art consequence analysis software [23], including the following elements:

- Calculate total effective dose equivalent (TEDE) for cloud, inhalation, ground, and resuspension for the plume exposure EPZ.
- Assume that ad hoc protective actions are taken beyond the EPZ boundary. These ad hoc actions would involve relocating people from regions outside the EPZ as appropriate. The timing of these protective actions is discussed further below.
- Develop MACCS input parameters for the applicable site and design-specific source terms. The SOARCA analyses for Peach Bottom and Surry (see reference [24], Appendix B and reference [25], Appendix C, respectively) provide examples for the default values of non-site specific and non-design specific input which were used for large LWRs and should be evaluated for applicability to SMRs.

Note that the EPZ boundary consequence calculation methodology proposed for Criterion a. differs from that used for offsite dose in Chapter 15 analyses in that it is based on the methodology typically used in severe accident dose calculations (see left hand lavender PRA box of Figure 4), for example, state-of-the-art severe accident consequence analysis software such as MACCS2. The methodology typically used for offsite dose in Chapter 15 utilizes atmospheric dispersion models described in Regulatory Guide 1.145 [26] which is not appropriate for EPZ boundary dose analysis for several reasons. First, the Regulatory Guide 1.145 methodology is limited to exclusion area boundary dose over a 2 hour period and outer low population zone boundary dose over a 30 day period, neither of which is applicable for EPZ boundary dose over a period which is sufficient for ad hoc protective actions. Second, the Regulatory Guide methodology leads to a highly conservative determination of dispersion factors instead of best-estimate values with uncertainty distributions that are appropriate for calculation of doses at the EPZ boundary. Third, the regulatory guide methodology has simplifications (e.g., cloud and inhalation exposures only (no ground shine), no particulate fallout, and no decay during plume transit from source to receptor) which are not appropriate for an EPZ boundary dose calculation that should be more realistic. In addition, it is desirable to have consistency with the severe accident consequence calculation methodology to be used for Criteria b. and c.

On the matter of the time required to implement ad hoc protective actions beyond the EPZ boundary, the SOARCA study, published in 2012, modeled relocation of the population outside the EPZ. Per Section 6.2.1 in both references [24] and [25], normal relocation was assumed to occur within 24 hours for the Peach Bottom site after plume arrival from locations where doses are projected to exceed 1 rem. Hotspot relocation for Peach Bottom was assumed to occur within 12 hours after plume arrival from locations where doses are projected to exceed 5 rem. For Surry, the times were 36 and 24 hours, respectively. These were average times and were based on SOARCA project review of emergency response timelines and the amount of time necessary for response personnel to identify the involved area and notify residents in this area that relocation is necessary, and for the residents to remove themselves from the area.

On the other hand, the EPA PAG Manual [11], published in 1992, indicates that the projected dose for comparison to the PAG is for exposure during the first four days following the start of release, and that the 4 days was chosen based on the time needed to make measurements, reach decisions, and prepare to implement relocation. The 4 days is also included in the 2013 PAG Manual draft for interim use and public comment.

It is apparent that the relocation times are site dependent, and that the SOARCA site-specific estimates are considerably shorter than the EPA estimate. It is also the case that, given modern technology such as GPS devices, remote monitoring, and in-situ monitors with real time data transmittal, it is reasonable to assume that measurements would be made promptly - within a few hours. Once decisions on protective actions were made, notifications using traditional methods plus more modern approaches such as text messaging, Internet, and social media could also be accomplished quickly. Relocations would be expected to take from a few hours to 12 hours based on typical evacuation time estimates. Thus the projected dose for comparison to the PAG could apply an exposure based on site-specific estimates of relocation time which consider the time needed to make measurements, reach decisions, and implement relocation. Modern technology available as part of the onsite and offsite emergency plans should be considered in such estimates. In the absence of a site-specific estimate, a default exposure of 4 days following the start of release should be applied.

Comparison of Dose with PAGs: The EPZ should encompass those areas in which projected dose from DBAs could exceed the PAGs. The EPA PAG dose values range from a low end of 1 rem to a high end of 5 rem [11]. Regarding confidence limits for comparison of the dose with the PAGs, it is proposed that the mean dose be less than 1 rem TEDE and that the 95% dose be less than 5 rem TEDE. This is consistent with the scalable EPZ approach discussed in conjunction with Table 1 in SECY-11-0152 where, if the "expected offsite dose" is less than 1 rem at a given distance, the requirements for the EPZ would be limited to a zone bounded by this distance. This also provides confidence that for more extreme, low probability meteorological conditions, the dose still would not exceed the 5 rem PAG at this distance.[4]

3.4 Methodology for Implementing Criterion b: "The EPZ should encompass those areas in which consequences of less severe core melt accidents could exceed the PAGs" (Part i)

The proposed methodology for implementing Criterion b., which is also in Part i of Figure 4, has the same four elements as for Criterion a. These elements are discussed below.

Accident Scenario Selection Criterion b. accident scenario selection is more involved than for Criterion a. since it uses frequency to inform selection of scenarios. The accident scenario selection uses PRA input as shown in the left hand PRA box on Figure 4.

[4] It is noted that the International Atomic Energy Agency (IAEA) specified dose for evacuation is also 5 rem (50 mSv) [27].

The process described below is one acceptable way that accident scenario selection can be accomplished using PRA. It is similar to the process used in SOARCA [10] but has been adapted for SMRs and expanded to be more complete. SOARCA was of course focused on operating plants, but the SOARCA accident scenario selection process is a recent and visible example of using PRA information which can be adapted by SMR applicants based on their respective plant-specific PRAs.[5]

The selection process from SOARCA for "most scenarios" was as follows:

1. Initial selection: Select all accident sequences with mean CDF > 1E-8 per reactor year.
2. Sequence evaluation: Evaluate dominant cut sets for the > 1E-8 per reactor year sequences. Determine system and equipment availabilities and accident sequence timing.
3. Scenario grouping: Group sequences with similar timing to core damage and similar equipment availabilities into accident scenarios.[6]
4. Scenario selection: Select accident scenarios with mean CDF > 1E-6 per reactor year.

Additional considerations to adapt this process for more probable, less severe (criterion b.) SMR accident scenario selection are as follows:

- For SMRs, this would need to be per plant year instead of per reactor year to support consideration of modular designs.
- In addition to scenarios with mean CDF > 1E-6 per plant year, all intact containment severe accident scenarios, even if below 1E-6 per plant year, should be included. This will assure that there are core damage scenarios to be evaluated for dose versus distance even with the low severe accident frequencies expected for SMRs.
- Basemat melt-through accidents (versus atmospheric release accidents) which were considered in reference [8] should also be included unless they are precluded by design.
- For simplicity, credit for operator mitigation actions would be limited to emergency operating procedures (EOPs), similar to the SOARCA unmitigated accident scenario evaluations. Severe accident management guidelines (SAMGs) and extreme damage

[5] As noted in the text, the accident selection and grouping process described here is taken from SOARCA [10] as is the term "accident scenario" i.e., a group of accident sequences with similar times to core damage and similar equipment availabilities. Other approaches to, and terminology associated with, accident sequence selection and grouping could also be used for SMR EPZ (see, for example, functional sequences and systemic sequences as defined in reference [28]).

[6] Care would be required in the grouping of accident sequences. There are several precedents for this (references [10] and [29]) where the general idea is to create a handful of scenarios, as opposed to a large number, so as not to have scenarios which are subdivided too finely (could reduce the frequency too much) and so that the number of scenarios requiring Level 2 analysis and dose calculation is more manageable.

mitigation guidelines (EDMGs) are considered in conjunction with the operationally-focused mitigation strategy for SMRs discussed in Section 4.[7]

- This set of scenarios is intended to encompass the more probable, less severe core damage scenarios. Thus, extreme seismic and other external hazards would not be part of this set of scenarios but rather will be considered under Criterion c. (Part ii of Figure 4). While to be confirmed on a design-specific basis, it is also expected that this set of scenarios would not involve impact on reactor modules from common or shared systems due to the very low frequency expected for such effects.

Accident Source Term Evaluation It is recommended that the proposed Criterion b. accident source term evaluation methodology use state-of-the-art, fully integrated, engineering level severe accident analysis software, such as MELCOR or MAAP5, and be PRA-based. The approach includes the following elements:

- Application of Level 2 PRA results (see left hand lavender PRA box on Figure 4)
- Development of a design-specific severe accident analysis model for performing the Level 2 analyses
- As discussed above, credit for operator mitigation actions would be limited to EOPs.
- Taking credit for installed safety and non-safety structures, systems, and components for accident interdiction and mitigation, consistent with their availability from Level 1 PRA results
- Calculation of fission product release to the environment versus time

EPZ Boundary Consequence Evaluation The EPZ boundary consequence calculation methodology is the same as that for Criterion a. and should be based on the methodology typically used for severe accident dose calculations (see left hand lavender PRA box on Figure 4).

Comparison of Dose with PAGs The methodology for comparison of dose with the PAGs including the confidence limits is the same as that for Criterion a.

3.5 Methodology for Implementing Criterion c: "The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe core melt accidents" (Part ii)

The elements of the proposed methodology for implementing Criterion c. (see Part ii of Figure 4) are as follows:

- Accident scenario selection (top light blue box on Figure 4)

[7] It is not intended to quantify probabilities for these additional operator mitigation actions in the Level 1 PRA since this would require a human reliability analysis to quantify operator success in carrying out mitigation actions, and methods for determining human error probabilities for accident management situations have not yet been proven.

- Accident scenario source term evaluation methodology, where source term in this context is the fission product release to the environment as a function of time (second light blue box on Figure 4)
- EPZ boundary consequence calculation methodology (also second light blue box on Figure 4)
- Comparison of consequences against early severe health effects (third light blue box on Figure 4)

Each of these elements is discussed below.

<u>Accident Scenario Selection</u> Like Criterion b., accident scenario selection for Criterion c. is PRA-based (see right hand lavender box on Figure 4) and is similar to the process used in SOARCA, but has been adapted for SMRs and expanded to be more complete. Consistent with reference [1], it will address the potential impact on reactor modules that have some common or shared systems. In addition, this supports the sizing rationale of consideration of a spectrum of accidents. Credible scenarios which involve extreme seismic and other external hazards are to be considered since such scenarios can contribute to risk. Finally, FHAs and SFP accidents are to be considered although for LWR SMRs it is expected that core damage accidents will be most limiting for offsite dose associated with plume exposure EPZ.

For accidents known to have the potential for higher consequences, SOARCA selected scenarios with mean CDF > 1E-7 per reactor year.[8] This would be changed to per plant year for SMRs to account for potential common cause effects on modules having common or shared systems. In performing the accident sequence grouping as part of scenario selection, accident sequences with mean CDF > 1E-8 per plant year should be considered in the initial sequence selection. As with Criterion b, credit for operator mitigation actions is expected to be limited to EOPs.

An additional consideration for Criterion c. accident scenario selection is that of extreme seismic and other external hazards. It is proposed that for external hazards, best-estimate initiating event frequencies approximately an order of magnitude below the frequency corresponding to the design basis be considered. The basis for this approach is that it puts the focus on hazards that more realistically could threaten the plant. If, for example, the safe shutdown earthquake were ~3E-4 per year for a given design at a given site, earthquakes with best-estimate initiating event frequency range down to ~3E-5 per year would be considered. To avoid penalizing SMR plants that have eliminated a given hazard by siting and/or design, or where the initiating event frequency for a given hazard is so low as to essentially eliminate risk from the hazard, it would be appropriate for the applicant to adjust this approach accordingly. In the case of seismic risk, for DC applicants a PRA-based seismic margin analysis (SMA) in accordance with Interim Staff Guidance DC/COL-ISG-020 [30] can be considered.

[8] Since Level 2 models and analyses are expected to be available, the selection could be based on radionuclide release frequency (e.g., large early release frequency (LERF)) rather than CDF. However, severe accident dose calculations will be performed which allows estimates of risk significance and makes LERF unnecessary. Also, bypass accidents, for which CDF represents radionuclide release frequency, are expected to dominate the less probable, more severe scenarios.

Consideration of hazards for lower initiating event frequencies is discussed in Section 4 as part of enhanced plant capabilities to account for uncertainties. Consideration of concurrent hazards such as correlated hazards is also addressed in Section 4.

The set of accident scenarios with mean CDF exceeding 1E-7 per plant year is intended to encompass the less probable, more severe core damage scenarios. Use of 1E-7 per year frequency as a basis for sequence selection has a number of precedents. These precedents were generally intended on a per reactor year basis whereas the 1E-7 mean CDF in this white paper is per plant year. The precedents include:

- NUREG-1338 [31] and NUREG-1420 [32] used a cutoff of 1E-7 per year.
- NUREG-1150 [29] used a frequency cutoff of 1E-7 per year for PRA accident sequence progression.
- Regulatory Guide 1.174 [12] specifies that an increase of 1E-7 per year in large early release frequency is permitted for proposed plant design changes.
- NUREG-0396 [2], Figure I-11, has a conditional probability range down to 1E-3 which corresponds to ~1E-7 per year absolute frequency.
- The lowest frequency considered in NUREG-1860 [33] was 1E-7 per year. NUREG-1860 also states in Volume 3, Section C.3.7 that, "… the Framework would require EP to consider accident sequences down to a frequency of 10^{-7}/ry, but no lower."

Industry recognizes that there are uncertainties in frequency quantification and that it is difficult to identify all potential safety challenges for a given design or site. Reference [33] discusses this issue in some detail, referring to "completeness uncertainty" which results from unknown and unforeseen failures or events for which it is difficult to estimate the magnitude of the uncertainty or the associated risk. This is a key reason for addressing additional steps to account for uncertainties (Part iii of Figure 4) which is discussed in Section 4.

<u>Accident Source Term Evaluation</u> As with Criterion b., it is recommended that the proposed Criterion c. accident source term evaluation methodology use state-of-the-art, fully integrated, engineering level severe accident analysis software, such as MELCOR or MAAP5, and be PRA-based (see right-hand box on Figure 4). It would also include the elements from the Criterion b. methodology. In addition, if a credible accident scenario involving more than one module were to be identified, equipment availabilities, core damage progression, and release timing would not be expected to be the same from one module to another. Thus the source terms and associated dose would not be expected to be additive, and this will need to be taken into account in the evaluation.

<u>EPZ Boundary Consequence Evaluation</u> The EPZ boundary consequence calculation methodology should be based on that typically used for severe accident dose calculations (right-hand box on Figure 4) similar to that for Criteria a. and b. except the relevant consequence parameter for Criterion c. is to provide substantial reduction in early severe health effects. Per

the NUREG-0396 framework, this is represented by the probability of dose exceedance for a dose that could cause early severe health effects. For a given source term and site, the probability of dose exceedance tends to decrease as distance from the reactor increases. This decrease in probability of dose exceedance with distance results from the fact that fewer and fewer weather trials (multi-hour weather sequences) occur in which the plume is stable enough to cause fission product concentrations that would result in a dose that could potentially result in early severe health effects. Specifics of the methodology for determining the probability of dose exceedance will need to be defined as part of implementation including possible updates to the approach described in NUREG-0396.[9]

<u>Comparison against Early Severe Health Effect Risk</u>   As stated in Section 3.2, Criterion c. specifies that the EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe core melt accidents. Figure I-11 of NUREG-0396 illustrates how this criterion was interpreted and implemented in 1978 as part of development of the basis for the existing 10-mile generic EPZ for current plants. In Figure II-11 of NUREG-0396, the probability of exceeding a whole body acute dose of 200 rem (taken in NUREG-0396 to be the dose at which significant early injuries start to occur) drops below about 0.01 beyond 10 miles from the reactor and declines rapidly to 0.001 beyond about 15 miles. Here, the probability is conditional on a core melt accident having occurred.

For purposes of the technical basis for SMR EPZ size, industry proposes a similar criterion to that in NUREG -0396, i.e., that the probability of exceeding a whole body acute dose of 200 rem drop in a rapidly declining manner at a distance approximately that of the EPZ boundary (or less). If conditional probability were used for a given design, the steps would be:

- For each of the selected scenarios for Criterion c., calculate the probability of exceeding 200 rem whole body acute dose as a function of distance (based on plume dispersion and thus radionuclide concentration in the plume being a function of distance)
- For a given distance, sum the scenario frequency-weighted probabilities over all scenarios
- Normalize (divide) by total CDF
- Plot the normalized sums vs. distance and determine the distance at which the result drops below 1E-3

At least some SMR designs are expected to have very low total CDF in which case applicants may want to use absolute probability instead of conditional probability (provides a better representation of risk). In any event, the applicant must demonstrate that the EPZ is of sufficient size to provide for substantial reduction in early severe health effects in the event of more

[9] NUREG-0396 used whole body acute dose, three exposure pathways (24 hour exposure to cloud shine, one year exposure to inhalation, and 24 hour exposure to ground shine), shielding factor of 0.7 for ground shine dose, no shielding factor for cloud shine dose, and no inhalation protection factor for inhalation dose.

severe core melt accidents, with the expectation that the probability of exceeding a whole body acute dose of 200 rem will drop rapidly at a distance approximately equal to or less than the EPZ boundary.

3.6 Base PRA Technical Adequacy and Uncertainty Evaluations

The applicant proposing appropriate EPZ sizing will need to establish that the technical adequacy of the base PRA is sufficient to support the application to determine EPZ size. Regulatory Guide 1.200 [6], which in turn refers to ASME/ANS RA-Sa-2009 [34], is one way to accomplish this. Used in support of the EPZ application, Regulatory Guide 1.200 should help focus the NRC review on key assumptions and areas identified as being of concern and relevant to the application.

Regulatory Guide 1.200 provides guidance on full-scope Level 1 PRA and limited Level 2 PRA sufficient to evaluate LERF. The framework outlined in this white paper does not require a Level 3 PRA but rather specifies offsite dose calculations which support evaluation against the proposed EPZ sizing criteria. The proposed methodology for EPZ boundary consequence calculation refers to the SOARCA analyses [24, 25] for examples of non-site and non-design specific input which can be evaluated for applicability to SMRs. The NRC is developing a full-scope, Level 3 PRA which will reflect current state-of-practice methods, tools, and data, and will incorporate technical advances since the last NRC-sponsored Level 3 PRAs which were completed over 20 years ago [35]. If appropriate, the results of this work can be factored into applicant EPZ-related dose calculations.

Uncertainty analyses should be performed as part of the base PRA. The need to identify and characterize the sources of uncertainty and potential sensitivities of the results related to assumptions and approximations is specified in Regulatory Guide 1.200, including assessing the impact of parameter uncertainties on the results and addressing each hazard group and its unique sources of model uncertainty.

NUREG-1855, Revision 0 [14], published in 2009, provides guidelines with regard to how to do this identification and characterization of the different sources of parameter and model uncertainty. Revision 1 of NUREG-1855 was issued for public comment just prior to the time of this writing. A January, 2013 ACRS letter [36] documenting a review of proposed Revision 1 indicates that it provides valuable guidance for treatment of uncertainties in risk information used for decision-making.

Application of the EPZ sizing methodology is an opportunity to provide a less burdensome, more predictable PRA adequacy process that is consistent with the real value of peer review, i.e., assessing of the degree to which the models realistically reflect the key plant-specific contributors to risk and the appropriateness of assumptions related to key areas of uncertainty. In this regard, the applicant should consider how the development of operator mitigation strategies discussed in Section 4 could support more practical approaches to the base PRA uncertainty analysis that may provide an alternative to problematic analytic treatments of uncertainties for very low probability events inside the PRA which may not be meaningful. An example of this is addressing completeness uncertainty for the SMR EPZ sizing application in Section 4 as part of additional steps to account for uncertainties.

3.7 Cumulative Plant Risk Design Objectives Quantified by PRA

The PRA-based evaluation discussed above addresses individual accident scenarios. It is also necessary to assure that the total plant risk does not exceed appropriate objectives. The PRA should be used to demonstrate that the following plant risk design objectives associated with the expectation of increased accident prevention and mitigation for SMRs are met for internal and external events and plant operating states amenable to PRA:

- Total mean CDF < 1E-5 per plant year
- Mean LERF < 1E-6 per plant year

As part of this demonstration, the applicant should determine the level of uncertainty in CDF and LERF, and evaluate the uncertainties against appropriate acceptance values for the uncertainties. The acceptance values are expected to be a function of the margin between the achieved mean CDF and LERF for the plant and the plant risk design objectives. The acceptance values should also factor in the smaller core power for SMRs, recognizing that core damage in a core that has significantly smaller thermal power than that of large LWRs likely does not have the same potential for radioactive release or impact on health and safety. This evaluation should then be used to define additional measures, if any, to be applied to the design or operation of the plant in order to reduce the uncertainties to acceptable levels.

4 Additional Steps to Account for Uncertainties (Part iii)

Section 4 discusses additional steps, in the form of enhanced plant capabilities to account for uncertainties (Part iii of Figure 4). As noted in Section 1 this is a complement to the PRA-based evaluation, and in large part is a deterministic, defense-in-depth approach. There are four steps to be performed by the applicant in this regard as discussed below.

The methodology proposed herein for enhanced plant capabilities to account for uncertainties is intended to be part of an integrated, decision-making process for SMR EPZ sizing which uses risk informed judgment to achieve a balance between use of quantitative, PRA information and application of a deterministic, defense-in-depth perspective, and which avoids the imposition of arbitrary and extreme accident sequence characteristics as part of the EPZ sizing basis.

4.1 Completeness Uncertainty Including an Operationally-Focused Mitigation Strategy

To address completeness uncertainty associated with the accident scenarios and source term, it is proposed that the applicant develop a diverse and flexible, operationally-focused strategy addressing both accident prevention and mitigation, on a design-specific basis. This strategy would provide an additional accident mitigation capability (beyond the installed plant systems and structures) that is based not on specific accident sequences nor on probabilities but rather on maintaining basic safety functions in the face of extreme site-wide situations, including potential impacts on reactor modules that have some common or shared systems, where it is difficult to foresee all potential conditions in advance. The basic safety functions would include core cooling, electric power, containment integrity, and spent fuel pool cooling under conditions of duress such as permanently installed equipment being unavailable and/or the site environs being damaged with limited access. If the strategy involves use of onsite portable equipment, that equipment would need to be dispersed and protected, and operators would require procedures and training.

In conjunction with developing such a system, the applicant should also do the following:

- Consider the need for pre-positioned, regional assets which are stationed in secure, offsite support centers and configuring the design to facilitate application of such assets
- Develop SAMGs and EDMGs and perform deterministic modeling of these mitigation strategies in the Level 2 PRA as was done for the SOARCA mitigated scenarios. The reliability of operator actions associated with these mitigation strategies should benefit from the simplicity of SMR designs, the expected slower fission product release, and the fact that the operator mitigation actions can be considered as part of the plant design versus after the fact in current plants.
- Show that there is sufficient time before the beginning of fission product release to the environment to allow operator mitigation strategies to be implemented and offsite protective actions to be accomplished. In this regard, applicants are expected to take advantage of SMR simplicity by keeping the mitigation actions simple with enough time to accomplish them.

An additional action to address completeness uncertainty is to show that detailed planning within the SMR EPZ provides a substantial base for expansion of response beyond the EPZ boundary. This is Part v of the Figure 4 EPZ approach which is to be addressed in the future.

4.2 Potential Risks that are Difficult To Quantify or Not Fully Addressed in the PRA

Examples of potential risks that may be difficult to quantify or not fully addressed in the PRA, and thus may need to be treated outside the PRA, are: security events, collateral damage, potential common cause effects on modules having common or shared systems, co-location, organizational performance, aging effects, factors affecting operations (e.g., shift staffing, training and procedures, use of new I&C systems, and lack of operating experience), errors of commission, design faults, risks treated outside the PRA due to an endorsed standard not being available as mentioned in the introduction to Section 3., and concurrent hazards as mentioned in Section 3.5.

To address this, it is proposed that applicants perform an engineering assessment (qualitative or quantitative as appropriate) to either: show that a given risk is adequately treated inside the PRA; or to confirm the existence, functionality, and capability of features and processes in the design, operation, accident management, and emergency response to address this risk and to provide confidence that the risk impact is acceptably low. The applicant will need to make the determination of what constitutes an acceptably low risk. It is noted that the decision of what is acceptably low risk may be based on qualitative factors and not based on quantitative, probabilistic metrics [13]. Examples of such features and processes include:

- Evaluation of the margin beyond the design basis for extreme external hazards and concurrent hazards such as source correlated hazards (e.g., seismic plus tsunami), phenomenologically correlated hazards (e.g., strong winds plus heavy rain), and induced hazards (e.g., seismically induced fire or flood) [16, 37].
- Using PRA insights together with a defense-in-depth deterministic approach as identified in the NEI SMR security position paper [38], demonstrate the robust nature of design for security events with margin beyond the design basis threat
- Use of a plant simulator as an integral part of the design process to validate the control room layout, operating procedures, shift staffing levels, and onsite emergency response
- The existence of a design-specific, plant-specific, flexible and diverse, operationally-focused accident mitigation capability as noted above in the discussion on completeness uncertainty in Section 4.1.

In the case of security, the recently proposed NRC RMRF contained in NUREG-2150 [13] states that the risk framework should be implemented for both safety and security-related issues and that "the NRC should have as a goal managing the appropriate amount of security defense-in-depth and better integrating security vulnerability assessments and risk assessments for other safety issues." At this point in time, however, methods are not well-established for estimating the risk of security-initiated events, hence the need for an engineering assessment and an operationally-focused mitigation system as suggested above.

4.3 Potential Impact on Risk of Lower Frequency Accident Sequences (Cliff Edge Effects)

Uncertainties in quantification for very low frequency accident sequences make it prudent to evaluate the potential impact on risk of these sequences. The applicant should accomplish this by performing the following evaluations:

- Extend the accident sequence frequency associated with Accident Scenario Selection described in Sections 3.4 and 3.5 to lower frequencies. Also, assess the potential for cliff-edge effects which could cause the risk to significantly exceed that from the selected accident scenarios that make up Parts i and ii of Figure 4. Note that the potential for significantly increased risk due to increased consequences will tend to be offset by lower frequency, as was noted in the SOARCA report [10].
- In lieu of extending the mean CDF down, perform the scenario selection with higher percentile accident sequences. For example, in the SOARCA-like process, this could involve selecting all accident sequences with 95th percentile CDF greater than ~1E-8 per

plant year and assess the potential for cliff-edge effects for accident scenarios with frequency below 1E-7 per plant year.

- Assess the CDF represented by selected accident sequences to confirm that a substantial fraction of total CDF is being addressed in the scenarios being evaluated.

- Extend the initiating event frequency for extreme seismic and other external hazards in Section 3.5 to lower frequencies. The specifics of such evaluations will depend upon the type of external hazard, the site under consideration, and the capabilities of the design. In the case of seismic, for example, an approach is described in an ICAPP 2010 paper [39] which suggests that initiating event frequencies no lower than 1E-5 per year be addressed, so as to avoid setting "a stringent goal that is overly burdensome and would force a licensee to focus on incredible events rather than on more likely threats to the power plant." Section D.1 of Regulatory Guide 1.208 [40] states that, "While the ASCE/SEI Standard 43-05 … approach only requires calculation of ground motion levels with mean annual probabilities of 1 E-04 and 1 E-05, NRC confirmatory activities also specify calculation of ground motion levels with a mean annual probability of 1 E-06." Regulatory Guide 1.200 indicates that for new reactor designs with substantially lower risk profiles (e.g., internal events CDF below 10^{-6} per year), the quantitative screening value should be adjusted according to the relative baseline risk value.

- On the other hand, the paucity of data for earthquakes and other external hazards at 1E-5 to 1E-6 would appear to make quantitative risk evaluations at such low frequencies impractical and/or not very meaningful. It is also noted that the development of an operationally-focused mitigation strategy which is based on maintaining basic safety functions regardless of the threat rather than being based on specific accident sequences or on probabilities, provides an alternative for addressing extreme seismic and other external hazards when quantitative risk evaluations are impractical due to lack of data at very low frequencies.[10]

In addition to the above, as suggested in reference [4], the applicant could confirm the existence of plant design capabilities, safety margin, and accident management capabilities which support the very low probability of sequences which were not selected and the low risk impact of these sequences due to any increased consequences being offset by lower frequency. In particular, historically important higher consequence, impaired containment accident scenarios in past LWR PRAs (e.g., interfacing LOCA, induced steam generator tube rupture) could be considered.

[10] It is noted, as stated in the SOARCA report, that in the case of very large, low frequency earthquakes, "…it would not be sufficient to perform a nuclear plant risk evaluation of this event without also assessing the concomitant nonnuclear risk associated with such a large earthquake. This assessment would have to include an analysis of the impact on public health of an extremely large earthquake…to provide the perspective on the relative risk posed by operation of the plant."[10]

4.4 Balance between Accident Prevention, Accident Mitigation, and Protective Actions

To assure reasonable balance between prevention, mitigation, and protective actions (an essential property of defense-in-depth and a further step in accounting for uncertainties), the applicant should assure the following:

- Both an onsite and an offsite emergency plan would be provided, including a certified offsite all hazards plan, which is based on the appropriately sized EPZ.

- Though not addressed in this paper, planning elements (including, for example, defining emergency action levels, drills and training, protective action strategies, and a modern public alert system) should be included in the emergency plan, and decisions on these planning elements should be integrated with the decision on the EPZ. (This is Part iv of Figure 4.) Also, the emergency plan should be consistent with the expectation of providing a substantial base for expanding response efforts should expansion be necessary. (This is Part v of Figure 4, also not addressed in this paper.)

- As provided by the methodology and criteria in this white paper, and in a similar way to the NUREG-0396 EP objectives for existing plants, a spectrum of accidents should be addressed. The size of the EPZ should be such that consequences from more probable, less severe accidents would not exceed the PAGs outside the EPZ, and should also provide for substantial reduction in early severe health effects in the event of less probable, more severe accidents.

5 Future Steps for Developing SMR EP

This paper provides a proposed methodology and criteria for establishing the technical basis for SMR plume exposure EPZ sizing and is intended to be the starting point for discussions leading to agreement that the methodology and criteria are a reasonable approach to guide the preparation of design certification and plant-specific license applications. It is anticipated that actual SMR applications requesting appropriate EPZ size and EP planning elements will be submitted at the combined operating license and/or early site permit stages (or the operating license stage under Part 50). Design certification applications, while not the licensing vehicle for defining plant-specific and site-specific EPZ size, will contain a significant amount of the technical information necessary to implement the methodology and criteria.

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Appendix A

Regulatory Background

Discussion of regulatory background provides context for industry's interest in determining appropriate EPZ sizing and the proposed approach for establishing the technical basis for accomplishing this. Protective actions in response to a nuclear power plant emergency were initially required in the early 1960's within 10 CFR Part 100 which specifies that every site must have an exclusion area and a low population zone (LPZ).[11]

At the same time that the Atomic Energy Commission was finalizing the Part 100 siting requirements, the Federal Radiation Council (FRC)[12] was introducing the concept of Protective Action Guides (PAGs). [A-1] The FRC described the application of protective actions as[13]:

> "In providing guidance for protective actions applicable to radioactive contamination of the environment, the Council is concerned with a balance between the risk of radiation exposure and the impact on public well-being associated with the alteration of the normal production, processing, distribution, or use of food.
>
> It is recommended that the term "Protective Action Guide" (PAG) be used to indicate the projected dose at which the above balance is judged to occur for the general types of protective actions considered in this section. Thus, the Protective Action Guide serves as a basis for deciding when such protective actions are indicated."

In 1970, Appendix E to 10 CFR Part 50 was added to specify in more detail the information required in emergency plans. Originally, the intent was that the 10 CFR Part 100 LPZ and 10 CFR Part 50, Appendix E requirements would be sufficient for power reactor licensees to undertake emergency protective action planning activities in the vicinity of their sites. By the mid-1970's, however, implementation was assessed as being inconsistent between sites. In response, a joint NRC/EPA Task Force was chartered, the goal of which was to provide a more definitive set of clarifying guidance. The regulatory framework defining specific EPZ sizes for power reactors was an outcome of this joint Task Force effort, published as NUREG-0396 [A-2] in late 1978. Endorsing the concepts in the Task Force report, the NRC issued a policy statement in which the EPZ sizing basis was described [A-3]:

[11] The LPZ is defined as the area immediately surrounding a plant's exclusion area where there is a reasonable probability that appropriate protective measures could be taken in the event of a serious accident.

[12] The FRC was subsumed into the Environmental Protection Agency upon its formation in 1970.

[13] The initial consideration was for protection of the food supply (following similar considerations from atmospheric tests of nuclear weapons) but, as power reactor designs increased in size, consideration of short-term protective actions to be taken in response to an accident plume were added.

"The major recommendation of the report is that two Emergency Planning Zones (EPZs) should be established around light water nuclear power plants. The EPZ for airborne exposure has a radius of about 10 miles; the EPZ for contaminated food has a radius of about 50 miles. Predetermined protective action plans are needed for the EPZs. The exact size and shape of each EPZ will be decided by emergency planning officials after they consider the specific conditions at each site. These distances are considered large enough to provide a response base which would support activity outside the planning zone should this ever be needed."

This concept for formal EPZs was also brought into a proposed rulemaking that was initiated following the TMI accident [A-4]. During the comment review period on the proposed rule, licensees for several reactors raised a concern that the EPZ sizing basis was overly conservative in that it drew on conclusions from the WASH-1400 Reactor Safety Study [A-5] without consideration of plant size. As WASH-1400 focused on large LWRs and on conservatively defined large release scenarios, the licensees argued that adoption of a single EPZ size standard did not recognize the reduced risk from smaller LWRs and thus would unnecessarily penalize such reactors. Upon consideration, the NRC adopted a revised basis for small water cooled power reactors (less than 250 MW thermal) and for the Fort St. Vrain gas-cooled reactor. In letters to the licensees, the NRC stated its reconsideration of the generic, one-size approach, noting[14]:

"This conclusion is based on the lower potential hazard from these facilities (lower radionuclide inventory and longer times to release significant amounts of activity for many scenarios). The radionuclides considered in planning should be the same as recommended in NUREG-0396."

Differences in risk were also acknowledged for other types of facilities. The Supplementary Information accompanying the final rule[15] on emergency planning requirements described three categories of facilities for which EPZ sizes were to be established, the large and small LWRs (as described above) as well as consideration of risk for even smaller reactors and fuel facilities [A-6]:

"The potential radiological hazards to the public associated with the operation of research and test reactors and fuel facilities licensed under 10 CFR Parts 50 and 70 involve considerations different than those associated with nuclear power reactors. Consequently, the size of Emergency Planning Zones (EPZs) for facilities other than power reactors and the

[14] See NRC letters dated April 14, 1980 to Pacific Gas and Electric Company (Humboldt Bay Nuclear Power Plant), May 19, 1980 to Public Service Company of Colorado (Fort. St. Vrain), June 13, 1980 to Consumers Power Company (Big Rock Point), and June 13, 1980 to Dairyland Power Cooperative (LaCrosse).

[15] Section 50.47, added in 1980, contains requirements on 16 planning standards that onsite and offsite emergency response plans must meet. The NRC uses these standards in its "finding" that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

degree to which compliance with the requirements … will be determined on a case-by-case basis."

In conjunction with reviews during the 1980s and 1990s on evolutionary and advanced reactor designs, the NRC evaluated the technical criteria and methods associated with emergency planning requirements. SECY-97-020 [A-7], "Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors," describes the review efforts and includes a detailed discussion of the EPZ sizing rationale first described in NUREG-0396. The SECY also describes the advances in source term and severe accident data and summarizes industry submittals. The NRC staff concluded:

> "The staff recognizes that the industry has made a significant effort to make the evolutionary and passive advanced LWRs safer than current designs, and that changes to EP requirements may be warranted if the technical criteria for EP requirements were modified to account for the lower probability of severe accidents or the longer time period between accident initiation and release of radioactive material for most severe accidents associated with evolutionary and passive advanced LWRs.
>
> In order to justify these types of changes to the EP basis, the staff believes that several issues, which would require significant expenditure of staff resources, need to be addressed: (1) the probability level, if any, below which accidents will not be considered for EP, (2) the use of increased safety in one level of the defense-in-depth framework to justify reducing requirements in another level, and (3) the acceptance of such changes by the Federal, State and local agencies responsible for emergency planning.
>
> Because industry has not petitioned for changes to EP requirements for evolutionary and passive advanced LWRs, the staff did not dedicate the resources to fully evaluate these issues. The staff remains receptive to industry petitions for changes to EP requirements for evolutionary and passive advanced LWRs, but does not intend to dedicate further staff resources until such a petition is received."

Adjusting emergency planning requirements has continued to be discussed in multiple NRC policy papers since SECY-97-020.[16] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing" [A-8], which documented a draft framework for technology-neutral regulation, includes consideration of provisions for adjusted emergency planning requirements:

> "The Framework has evaluated existing EP requirements contained in 10 CFR 50.47 and 10 CFR 50, Appendix E in light of the defense-in-depth recommendations discussed in Section C.3.1. The defense-in-depth recommendations include retaining EP as a defense-in-depth measure, regardless of the plant design. In Appendix G, the Framework proposes an

[16] Notably, SECY-02-0139, SECY-03-0047, SECY-04-0157, SECY-05-0130, and SECY-10-0034.

> approach of retaining the 10 CFR 50.47 and 10 CFR 50, Appendix E requirements, but adding a provision that would allow future applicants to propose adjustments to current EP requirements based upon plant specific characteristics (e.g., timing of release, magnitude of release, plant risk). This approach would recognize that different plant characteristics may result in different EP needs and would permit applicants to propose appropriate adjustments (e.g., EPZ size, protective actions). Defense-in-depth and security would be key considerations in reviewing such proposals. In addition, other factors would need to be considered in reviewing proposed changes to EP requirements."

A 2011 NRC policy paper on EP adjustments is SECY-11-0152 [A-9], "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors." In discussing an approach for scalable EPZs, the SECY notes on page 3:

> "Although the guidance in NUREG-0396 and EPA-400 was written for large LWRs, the principle of using dose savings to determine EPZ size can also be applied to SMRs… With the expected safety enhancements in SMR designs and the potential for reduced accident source terms and fission product releases, the staff believes that it may be appropriate for SMRs to develop similarly reduced EPZ sizes, commensurate with their accident source terms, fission product releases, and accident dose characteristics."

and on page 4:

> "The staff considers it appropriate to be open to applicant requests for establishing SMR technology-neutral, variable distance, plume exposure EPZs"

and on page 6:

> "A scalable EPZ scheme would allow for regulatory predictability for SMR applicants and for State and local officials. This approach would ensure the consistent application of NRC regulations and requirements in the review of EP plans prepared for SMRs. This approach is consistent with current EP requirements and would not result in a reduction in the protection of public health and safety."

The most recent NRC paper addressing SMR EP is SECY-12-0139 [A-10], "Annual Update on the Status of Emergency Preparedness Activities." This paper summarizes the status of a more risk-informed and performance-based regulatory approach for existing plant EP programs with a focus on exercise scenarios, emergency action levels, and offsite programs. SECY-12-0139 also provides additional information on the EP framework for SMRs discussed in reference [A-9], stating:

> "In recent interactions, stakeholders have expressed an interest for the staff to develop a technology-neutral, dose-based EP framework that takes into account the SMR modular design and its collocation with industrial processes to determine the appropriate size of the EPZ. The specific areas of focus are staff positions regarding new policies or revised

regulations for the EPZ size, protective action guidelines, and guidance for a graded approach to specific 10 CFR Part 50 EP requirements. The staff informed the stakeholders that the NRC's existing regulatory structure provides the framework for the development of an emergency plan for an SMR. The staff informed industry that future EP work on SMRs will consider the various designs, modularity, and collocation, as well as the size of the EPZ, once an application has been received. The results of the staff's review of an SMR application could serve as the basis for regulatory revisions. The staff anticipates that applicants will file SMR design certification applications in the third and fourth quarters of 2013, and that the nuclear industry could separately submit EP position, topical, and technical papers for NRC review."

In summary, the regulatory background indicates that the existing EPZ sizing rationale (i.e., to produce dose savings for a spectrum of accidents) continues to be sound. In this regard, industry agrees with this principle and also believes that there is a need for updating the rationale as applied to SMRs to make it more risk-informed. The regulatory background supports the potential for adjusting emergency planning requirements for SMRs pending submittal of a formal proposal from industry.

Appendix A References

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